NON-PUBLIC?: N
ACCESSION #: 9301270052
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Pilgrim Nuclear Power Station PAGE: 1 OF 8

DOCKET NUMBER: 05000293

TITLE: Reactor Scram and Closing of Main Steam Isolation Valves
Due to Trip Settings of Main Steam Radiation Monitors
EVENT DATE: 12/20/92 LER #: 92-018-00 REPORT DATE: 1/18/93

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 75

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Engineer

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On December 20, 1992, at 0233 hours, an automatic Reactor Protection System (RPS) scram signal and scram, together with the closing of the Main Steam Isolation Valves, occurred while at 75 percent reactor power. Automatic responses included a transfer of electrical loads, Turbine-Generator trip, and closing of applicable isolation valves.

The root cause of the event was utility non-licensed Instrumentation and Control technician error. The technician incorrectly adjusted the trip settings of the Main Steam Radiation Monitors during power ascension on December 19, 1992. Hydrogen injection began on December 20, 1992, and the resulting increase in main steam radiation exceeded the trip settings that were too low for hydrogen injection radiation levels. Contributing to the error was the format of the settings identified in the trip setting procedure. The required settings were identified in a whole number format (e.g., 14,900) while the radiation monitors display the settings in a scientific notation format (e.g., 1.49E4). Corrective action taken included revising the procedure to identify the trip settings in scientific notation format and improved verification. The unit returned to commercial service on December 23, 1992, at 0622 hours. This event occurred during power operation with the reactor mode selector switch in the RUN position. The Reactor Vessel (RV) pressure was 1005 psig with the RV water temperature at approximately 545 degrees Fahrenheit. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) and this event posed no threat to the public health and safety.

END OF ABSTRACT

BACKGROUND

The Hydrogen Water Chemistry program uses the injection of hydrogen to decrease the rate of intergranular stress corrosion in recirculation piping to below detectable levels. The injection of hydrogen results in an increase of radioactive nitrogen and accompanying radiation increase in the main steam lines.

The safety objective of the Primary Containment and Reactor Vessel Isolation Control System is to initiate the automatic isolation of appropriate pipelines which penetrate primary containment whenever monitored variables exceed preselected operational limits. High radiation in the vicinity of the main steam lines could indicate a gross release of fission products from the fuel. The trip setting of the Main Steam Radiation Monitors RM-1705-2A/B/C/D is selected high enough above background radiation levels to avoid spurious isolation, yet low enough to promptly detect a gross release of fission products from the fuel. The trip settings are specified by Technical Specification Table 3.1.1 at 7 times normal full power background. Moreover, within 24 hours of the start of hydrogen injection with reactor power at >/= 20 percent of rated power, the trip settings are increased to
radiation levels expected during hydrogen injection.

Procedure 3.M.2-7.6, "NUMAC Log Radiation Monitor Setpoint Change Procedure", is used for adjusting (i.e., increasing or decreasing) the trip and alarm settings of RM-1705-2A/B/C/D, Main Steam Radiation Recorder RR-1705-11, and the Steam Jet Air Ejector (SJAE) Radiation Monitors RM-1705-3A/B. Recorder RR-1705-11 is normally set to alarm at a lower level than the trip settings of RM-1705-2A/B/C/D. The trip settings of RM-1705-2A/B/C/D and alarm setting of RR-1705-11 were adjusted by Instrumentation and Control (I&C) personnel on December 19, 1992, at approximately 2030 hours using procedure 3.M.2-7.6 (Rev. 2).

On December 20, 1992, operating conditions just prior to the event included the following. The reactor mode selector switch in the RUN position. The Reactor Vessel (RV) pressure was 1005 psig with the RV water temperature at approximately 545 degrees Fahrenheit. The RV water level was approximately inches. The Recirculation System motor generator sets/pumps 'A' and 'B' were in service with each loop in the manual control mode. Reactor core flow was approximately 37.7 million pound per hour. The Condensate System and Feedwater System pumps were all in service. The Feedwater Level Control System was in the three element control mode. The SJAE portion of the Main Condenser Gas Removal (MCGR) System was in service. The injection of hydrogen began at 0225 hours as part of power ascension.

EVENT DESCRIPTION

On December 20, 1992, at 0233 hours, an automatic Reactor Protection System (RPS) scram signal and scram, together with an actuation of the Group 1 portion of the Primary Containment Isolation Control System (PCIS), occurred while at 75 percent reactor power. The Group 1 actuation resulted in the following designed responses:

o The Primary Containment System (PCS) Main Steam Isolation Valves (MSIVs) closed automatically.

o The PCS/Main Steam drain piping isolation valves closed automatically.

o The PCS/Reactor Water sample isolation valves closed automatically.

The scram resulted in automatic responses that included a transfer of the source of 4160 VAC power for the Auxiliary Power Distribution System from the Unit Auxiliary Transformer to the Startup Transformer, trip of the Turbine-Generator, opening of 345KV switchyard ACBs 104 and 105, and runback of the Recirculation System motor-generator sets/pumps 'A' and 'B'. The scram also resulted in a decrease in the RV water level due to a decrease in the RV water void fraction. The RV water level ultimately decreased to approximately - 6 inches. The decrease to less than the RV low water level setpoint (calibrated at approximately inches) caused an automatic actuation of the PCIS and Reactor Building Isolation Control System (RBIS).

The PCIS responses included the following designed responses:

o The PCS Group 2/Sampling System isolation valves that were open closed automatically.

o The PCS Group 3/Residual Heat Removal (RHR) System isolation valves, in the closed position, remained closed.

o The PCS Group 6/Reactor Water Cleanup (RWCU) System isolation valves closed automatically.

The RBIS actuation included the automatic closing of the Secondary Containment System (SCS)/Reactor Building Trains 'A' and 'B' ventilation supply and exhaust dampers, and start of the SCS/Standby Gas Treatment System (SGTS) Trains 'A' and 'B'.

Control Room operator response to the event was orderly and included the following. The reactor mode selector switch was moved from the RUN position to the SHUTDOWN position and then to the REFUEL position as part of verifying the insertion of all control rods. The Feedwater System Trains 'A' and 'B' block valves were closed. Procedure EOP-01 (Rev. 1), "RPV Control", was entered because the RV water level was less than inches. At 0237 hours, the High Pressure Coolant Injection (HPCI) System was put into service in the full flow test mode for RV pressure control in accordance with EOP-01. The RHR System Loops 'A' and 'B' were put into service in the Suppression Pool Cooling mode at 0239 hours because of the addition of heat from the HPCI turbine exhaust steam. Procedure 2.1.7 (Rev. 27), "Vessel Heatup and Cooldown", was initiated. At 0244 hours, the Chemistry Division was notified to test for fuel damage. Only after determining the cause of the event and the completion of testing that revealed no fuel damage had occurred, the PCIS was reset at 0253 hours and the RWCU System was subsequently returned to service. EOP-01 was exited at 0300 hours when the RV water level returned to greater than inches. At 0323 hours, EOP-03 (Rev. 1), "Primary Containment Control", was entered because the Suppression Pool temperature was greater than 80 degrees Fahrenheit. At 0328 hours, ACBs 104 and 105 were reclosed with the main disconnects (T930) in the open position. At 0350 hours, the RPS was reset. After equalizing the MSIV differential pressure in accordance with procedure 2.2.92 (Rev. 25), "Main Steam Line Isolation and Turbine Bypass

Valves", section 7.1, the MSIVs were reopened at 0414 hours. The HPCI System was returned to standby service at 0419 hours. At 0445 hours, procedure EOP-03 was exited when the Suppression Pool temperature was less than 80 degrees Fahrenheit.

Problem Report 92.9285 was written to document the event. The NRC Operations Center was notified in accordance with 10 CFR 50.72 on December 20, 1992 at 0332 hours.

A post trip review was initiated in accordance with Procedure 1.3.37 (Rev. 7), "Post Trip Reviews".

CAUSE

A critique of the event was held on December 20, 1992, and was attended by appl
cable personnel including one of the I&C technicians who performed procedure 3.M.2-7.6 on December 19, 1992.

The root cause of the event was utility non-licensed I&C Technician error on December 19, 1992, when the trip settings of RM-1705-2A/B/C/D were adjusted using procedure 3.M.2-7.6 (Rev. 2). Contributing to the error was the method of recording the required settings prior to the adjustment. The required trip settings were identified in the procedure in a whole number format. During the adjustment of the monitors the trip settings were incorrectly entered. The as-left trip settings were recorded from the monitor's display (e.g., 1.49E3) and verified to be correct. The settings were not detected as incorrect (1.49E3 versus 1.49E4) during the verification. The alarm setting of RR-1705-11 was correct for hydrogen injection radiation levels. The completed procedure was left for subsequent I&C Supervisor review on the next working day (December 21, 1992).

The radiation monitors are vertically mounted and located in the Main Control Room at Panel C-910. The location is adequately illuminated and the monitors are approximately 4 to 5 feet above floor level.

CORRECTIVE ACTION

Procedure 3.M.2-7.6 was revised (to Rev. 3) to record alarm and trip settings in scientific notation and to specify the comparison of as-left settings to required settings.

The unit returned to commercial service on December 23, 1992, at 0622 hours. After a pre-evolutionary briefing at 2148 hours, the trip settings of RM-1705-2A/B/C/D were adjusted (i.e., increased) prior to hydrogen injection. The subsequent injection of hydrogen occurred without further incident. Full reactor power operation was achieved on December 24, 1992, at 0355 hours.

Subsequent followup of the trip settings of RM-1705-2A/B/C/D and alarm setting of RR-1705-11 was conducted by the NRC Resident Inspector. The followup identified the alarm setting of RR-1705-11 had not been decreased after the scram and prior to the resumption of hydrogen injection on December 24, 1992, at 0036 hours. Problem Report 93.9006 was written regarding the alarm setting of RR-1705-11 during that period. The cause and corrective action will be tracked via NRC Inspection Report 92-28.

The evaluation of human performance for the event was in progress when this report was prepared. Corrective action currently planned includes the evaluation of Procedure 3.M.2-7.6 for further improvement. Additional actions may be identified as a result of the human performance evaluation.

PREVENTIVE ACTION

The timeframe for I&C supervisory/management review of completed surveillance procedures is being evaluated as part of a previously identified corrective action program audit finding (DR 1998).

DISCIPLINARY ACTION

The responsible I&C technicians who performed procedure 3.M.2-7.6 on December 19, 1992 received disciplinary action.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The Core Standby Cooling Systems (CSCS) consist of the HPCI System, Automatic Depressurization System (ADS), Core Spray System, and the RHR/Low
Pressure Coolant Injection (LPCI) mode. The HPCI System provides water to the RV for high pressure core cooling. Although not part of the CSCS, the Reactor Core Isolation Cooling (RCIC) System is capable of providing water to the RV for high pressure core cooling, similar to the HPCI System. The ADS is a backup to the HPCI System and functions to reduce RV pressure to

enable low pressure core cooling provided independently by the Core Spray System and the RHR/LPCI mode. The CSCS and RCIC System were operable.

The lowest RV water level that occurred was approximately -6 inches. The level was greater than the level corresponding to the CSCS low-low water level setpoint (calibrated at approximately -45 inches) and the level (-127.5 inches) corresponding to the top of the active fuel zone.

The highest RV pressure that occurred was approximately 1020 psig. The pressure was less than the Main Steam relief valves' setpoint (1115 11 psig) and Main Steam safety valves' setpoint (1240 13 psig).

The RV water level was also greater than the setpoint (calibrated at approximately - 46 inches) that initiates the Anticipated Transient Without Scram (ATWS) System functions for a Recirculation Pump Trip (RPT) and Alternate Rod Insertion (ARI). The RV pressure was also less than the setpoint (calibrated at approximately 1175 psig) that initiates the ATWS System RPT and ARI trip functions and the setpoint (calibrated at approximately 1400 psig) that initiates the ATWS System function for a Feedpump Trip.

The highest Suppression Pool bulk water temperature that occurred was approximately 83 degrees Fahrenheit. The temperature was less than the maximum water temperature (120 degrees Fahrenheit) specified by Technical Specification 3.7.A.1.h during RV isolation conditions.

Technical Specification 3.7.A.1.m specifies the Suppression Pool/Chamber be maintained between -6 to -3 inches which corresponds to a downcomer submergence of 3.00 and 3.25 feet, respectively. The highest Suppression Pool water level that occurred was approximately -4 inches (129 inches on LI/LR-1001-604A/B). The level was less than the level corresponding to the maximum Suppression Pool volume of 94,000 cubic feet specified by Technical Specification 3.7.A.1.b. A Suppression Pool volume of 94,200 cubic feet corresponds to a level of inches (LR-5038/5049) or 139 inches (LI-1001-604A/B). The level was also less than the settings of the level switches (LS-2351A/B) that control the Suppression Pool/HPCI pump suction valves.

Control Room operator actions for response to the closing of any MSIV is addressed in alarm response procedure ARP-905R and procedure 2.4.30, "MSIV Closure".

The Main Condenser Gas Removal (MCGR) System functions to remove air, non-condensable gases and water vapor from the Main Condenser. The MCGR System includes the SJAE and Mechanical Vacuum Pump (MVP) subsystems. The
SJAE subsystem exhaust is metered, sampled, and monitored via SJAE radiation monitors RM-1705-3A/B prior to entering the offgas system holdup piping. Instrumentation connected to RM-1705-3A/B functions to isolate the offgas flow from discharge to the Main Stack if necessary. Each radiation monitor has an upscale and downscale trip function. The upscale trip indicates high radiation, and a downscale trip indicates instrument trouble. Any one trip provides an alarm in the Main Control Room at Panel C-903R. Any two trips actuates a time delay which causes the closing of drain valves and an outlet valve in the offgas piping. The MVP subsystem is operated during startup and shutdown when adequate steam pressure is not available to operate the SJAE, and the volume of air and gases exceeds the capacity of the SJAE. In addition to providing a Reactor Protection System

trip function, the Main Steam Radiation Monitors provide a trip function that causes a Mechanical Vacuum Pump trip and closing of a valve in the MVP piping to the Turbine-Generator Gland Seal holdup line that exhausts to the Main Stack. The Main Stack is equipped with a monitoring system that includes RM-1705-18A/B. Each monitor has two upscale trip functions and one downscale trip function. Each trip initiates an alarm at Control Room Panel C-903R. The upscale alarms indicate high radiation, and the downscale alarm indicates instrument trouble. The Main Stack monitoring system instrumentation is calibrated, checked, and functionally tested in accordance with procedures scheduled and tracked via the Master Surveillance Tracking Program. The Main Stack instrumentation was operable during the period the alarm setting for RR-1705-11 had not been decreased after the scram on December 20, 1992, and prior to the resumption of hydrogen injection on December 24, 1992. Control Room operator actions for response to a Main Stack high radiation condition are addressed in alarm response. procedure ARP-903R and procedure 2.4.40, "Rapid Increase in Offgas Activity".

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the actuation of the RPS and Group 1 portion of the PCIS, although designed responses to a high main steam radiation trip signal, were not planned.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) involving the trip settings of the Main Steam Radiation Monitors or similar event. The review identified LER 85-021-00 that involved the trip settings of the former Main Steam Radiation Monitors.

For LER 85-021-00, the trip settings of two of the former Main Steam Radiation Monitors were greater than specified by Technical Specification Table 3.1.1. The discovery occurred on August 14, 1985, while at 100 percent reactor power and as a result of review of the Chemistry Division Daily Log. The cause was utility nonlicensed personnel error. The error occurred because there was no approved procedure in place requiring appropriate action to be taken when the daily readings were found to be outside acceptable limits. Corrective action taken included adjusting the trip settings within specified limits and the addition of a note to the background radiation daily log sheet requiring Chemical Engineer notification when Main Steam Radiation Monitor readings decrease more than 125 mR/hr below posted background radiation levels. Procedure 7.4.31 (Rev. 0) was subsequently issued to provide instructions including notification of the Chemical Engineer and Watch Engineer if the trip settings were found out of acceptable limits. The former vacuum tube type log Main Steam Radiation Monitors were replaced with solid state type NUMAC log radiation monitors during the 1986-1988 outage.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS CODES

Monitor (RM-1705-2A/B/C/D) MON

SYSTEMS

Containment Isolation Control System (PCIS) JM
Engineered Safety Features Actuation System (PCIS, RPS) JE
HPCI System BJ
Main Steam System SB

ATTACHMENT 1 TO 9301270052 PAGE 1 OF 1

10 CFR 50.73

BOSTON EDISON

Pilgrim Nuclear Power Station
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E. Thomas Boulette, PhD January 18, 1993
Vice President Nuclear Operations BECo Ltr. 93-007
and Station Director

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 92-018-00, "Reactor Scram and Closing of Main Steam Isolation Valves Due to Trip Settings of Main Steam Radiation Monitors," is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

E. T. Boulette
Senior Vice President Nuclear (Acting)

DWE/bal

Enclosure: LER 92-018-00

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